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Guide to General Atomic Studies of Hypothetical Nuclear Driven Accidents for the Fort St. Vrain Reactor

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GUIDE TO GENERAL ATOMIC STUDIES OF HYPOTHETICAL NUCLEAR DRIVEN ACCIDENTS FOR THE FORT ST. VRAIN REACTOR

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GUIDE TO GENERAL ATOMIC STUDIES OF HYPOTHETICAL NUCLEAR DRIVEN ACCIDENTS FOR THE FORT ST. VRAIN REACTOR

Thomas Wei*   Melvin Tobias

Abstract

The work of the General Atomic Company (GAC) in preparing those portions of the Final Safety Analysis Report for the Fort St. Vrain Reactor (FSV) having to do with hypothetical nuclear driven accidents has been reviewed and a guide to this literature has been prepared. The sources for this study are the Final Safety Analysis Report itself, the Quarterly and Monthly Progress Reports, Topical Reports, and Technical Specifications. The problems considered and the methods used are outlined. An appendix gives a systematic analysis which was used as a guide in organizing the references.

Keywords: accidents, reactivity, reactor safety, safety, HTGR.

I. INTRODUCTION

The staff of Oak Ridge National Laboratory is engaged in HTGR Safety Studies on behalf of the Directorate of Licensing and for the Division of Reactor Research and Development of ERDA. The purpose of this review is to provide a guide to the problems considered and the methods used by the General Atomic Company in connection with the Final Safety Analysis report for the Fort St. Vrain reactor. This information is also basic to an understanding of the approach taken by GAC in connection with the nuclear driven safety problems of larger HTGR concepts.

The subjects of this document are then the following:

1) The safety problems considered by GAC in the preparation of the FSAR.
2) A guide to the sources used for physical data.
3) A guide to the calculational methods used.

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*This report summarizes work done by Mr. Wei, a graduate student at MIT, as a summer participant at ORNL during 1973.

II. SURVEY OF REACTIVITY RELATED SAFETY ANALYSIS STUDIES BY GAC FOR THE FORT ST. VRAIN REACTOR

Reactivity-related safety questions are discussed in two ways in the FSAR. One of the ways is to attempt to decide what the consequences may be of a reactivity addition which might occur. The rod withdrawal accident is typical of this line of thought. In the other approach, the reactivity effects of incidents not necessarily originating in a nuclear event are examined. The principal example of this kind of discussion is the permanent loss-of-forced circulation (LOFC) hypothetical accident.

The reactivity and kinetics safety questions discussed in the FSAR are outlined below and, whenever possible, background reference material is cited.

A. Excessive Removal of Control Poison

1. Rod-pair withdrawal during normal operation at various powers

Of the several reactivity incidents examined in the safety analysis reports, this is the most important and has been discussed almost continuously over the past six years in the quarterly reports for the Fort St. Vrain reactor.

GA-7086. Discusses transient analysis for maximum rate of withdrawal of highest worth control rod pair. Initial power 833 MW. Scram assumed at 140% power with 200 Msec delay or by coolant overtemperature scram at 1525°F with 15-sec delay. Power level and temperatures displayed as functions of time.

GA-7314. Transient analysis results reported for equilibrium cycle where delayed neutron fraction is least and temperature coefficient least negative. Analysis carried out for both high power and source power. Transient can be terminated by a 120% full-power rod withdrawal prohibit or by 140% overpower scram. Maximum fuel temperatures and gas outlet temperatures for hottest fuel elements and hottest channels reported.

GA-7453. Transient analysis described under improved protective conditions at source power, 25% full power, and full power. Entire problem outlined, giving initial conditions of temperature power and flow.

GA-7634. Asymptotic final gas outlet temperature estimates given for rod withdrawal accident as a function of flow rate, temperature coefficient, and rod worth.

GA-8038. Discusses sensitivity analysis of uncertainties in fluid flow and heat transfer model on transient power levels and temperatures calculated for a rod withdrawal accident. The parameters were the thermal conductivities of the kernel, buffer layers, high density isotropic layer and fuel bed, the core heat capacity, and the temperature coefficient.

GA-8420. Presents an "extended and improved" analysis of source power accident at 10% coolant flow, which supersedes analysis of GA-7453.

GA-8600. Sensitivity of startup rod withdrawal accident to prompt neutron lifetime shown to be small (FSAR gives lifetime of 2.6×10^{-4} sec; revised estimate given here is 4.3×10^{-4}). Rod worth said to be "surprisingly" sensitive to graphite density.

GA-9720. Results given for revised analysis of rod withdrawal accident. Used BLOOST point kinetics code as well as two-dimensional diffusion calculations. Sensitivity to control-rod worths obtained by using worths of 0.01, 0.016, and 0.025 Δk. Fuel temperature rises and average gas outlet temperatures computed for shutdowns initiated at 140% full power, 60 sec after accident start, and 105 sec after start.

GA-9875. Discusses sensitivity of results in rod withdrawal accident calculations to rod worth, temperature coefficient, and reactivity insertion rate.

GA-10313. Consequences of moving wrong set of control rods when adjusting reactivity described.

GA-10560. Source power rod withdrawal accident reviewed under more realistic (less conservative) assumptions. Average fuel temperature rise was 300°C and maximum rise in hottest channel was 1600°C.

GA-10850. Calculated consequences given for a rod withdrawal accident under end-of-cycle core conditions for 4%, 25%, and 100% rated power. Temperatures resulting from 140% power scram, 1075°F reheat steam temperature scram, and no scram, displayed for the average channel.

GA-A12477. Discusses calculations of transient behavior for rod withdrawal accident during zero-power startup experiment in air. Initial conditions are a power of 0.08 W, temperature 100°F, and flow at 0.1%. A scram occurs 50 sec after the start of the accident. Concludes under the restrictions of the calculation that little oxidation will take place.

2. Rod-break accident

The FSAR, Sect. 14.2, dismisses the importance of this accident on the grounds that average core temperatures would rise less than 150°C even without a scram. Essentially the same discussion is found in GA-8038.

3. Thirty-seven-rod-pair withdrawal accident

Withdrawal of more than one-rod pair accidentally is regarded as incredible, for the reason, as stated in FSAR 14.2, that the control system permits withdrawal of only one-rod pair at a time. However, the power level, average fuel temperature rise, and average gas outlet temperature rise from source and 100% power were calculated.

GA-8270. Gives temperatures and power resulting from a linear reactivity insertion of 0.00286 Δk/sec with a scram at 140% (1161 MW) of rated power for least negative, equilibrium cycle, temperature coefficients.

4. Rod-pair ejection accident

GA-7453. Discussion which concludes there is no event sequence that can produce this accident.

GA-8420. Discussion of rod-ejection accident giving description and results for fuel temperature rise and other consequences. Concludes that some fuel particle failures would occur.

B. Loss of Fission-Product Poisons

The FSAR (14.2.1.2) states that it is not possible to release noble gas poisons at intervals which are short relative to control shutdown times. This type of reactivity insertion "cannot lead to accident conditions as severe as those associated with excessive removal of control poison."

GA-8270, p. 80. Discusses the reactivity involved as part of the core heat-up accident (permanent loss of cooling accident). See also Section 14.10.3.3, Appendix D.1.2.3.2, and Appendix D.1.3.3.2 in FSAR.

C. Rearrangement of Core Components

The hazards described are adjudged much less severe than the rod withdrawal accident (FSAR 14.2.1.3) and no credible way was conceived in which a core could be accidentally rearranged to constitute a limiting reactivity hazard. The geometry changes investigated were a large accumulation of graphite in the core coolant channels, compression of the core volume (due to earthquake for instance), and addition of a fuel element in refueling. (No references for these matters besides the FSAR are known to us.)

D. Introduction of Steam into the Core

FSAR (14.2.1.4) states that the rate of reactivity insertion is so low that this accident has not been explicitly studied. The net coefficient of reactivity addition is given as 2.1×10^{-5} Δk/lb water.

GA-8600, p. 67 (March 31, 1968). Gives much the same discussion but somewhat different numbers are cited. An earlier calculation is mentioned but not referenced.

E. Sudden Decrease in Reactor Temperature

FSAR (14.2.1.5) states that this accident is less severe than the rod withdrawal accident and was not studied in detail. It is remarked that significant core temperature changes can only occur in a subcritical system with high coolant flow. The large heat capacity of the reactor makes quick changes in temperature impossible; their calculations show that if the entire normal power output were deposited in the core its temperature would rise less than 5°F/sec. Some additional discussion is to be found in GA-8038 and GA-8420. The maximum rate of reactivity addition is estimated in the FSAR as 6×10^{-5} Δk/sec.

F. Nuclear Consequences of the Loss-of-Forced-Circulation Accident (LOFC)

The LOFC receives extensive attention in the FSAR both in the main body of the report (14.10) and in the lengthy Appendix D (161 pages, 100 figures, 32 tables). The reactivity consequences are discussed in Sections 14.10.3.3, D.1.2.3, and D.1.3.3. (There are additional discussions concerning fission-product release and distribution which are outside the scope of the present review.) Section 14.10 concludes that reactivity removal on scram would remove 0.180 Δk (rods) plus 0.1 Δk (reserve shutdown material). The decay or escape of xenon and other fission-product poisons, the redistribution of thorium and uranium, control-rod compaction, and cooling of the core add an estimated 0.23 to the reactivity. Boron diffusion was neglected as a conservative approximation, but compaction and melting phenomena were taken into account.

GA-8270, p. 80. An older discussion of the same matter but much briefer.

III. CRITICALITY SAFETY QUESTIONS OUTSIDE OF THE REACTOR

The conclusion in all of the following is that adequate design has averted serious safety questions. We have confined ourselves here therefore to simply citing the references.

A. Flooding of the Fuel Storage Wells

GA-7453, p. 65.
GA-7939, p. 53.
GA-8270, p. 82.

B. Flooding of the Fuel Handling Machine

GA-7453, p. 65.

C. Miscellaneous Criticality Problems

1. Criticality of fresh fuel shipping container

   GA-9720, p. 29.

   GA-9875, p. 25.

2. Criticality of spent fuel shipping cask

   GA-9440, p. 40.

3. Criticality of fresh fuel element array

   GA-10313, p. 33.

See also GAMD-10493 concerning safe storage criteria for fuel under various conditions.

IV. SOURCES OF DATA NECESSARY FOR CRITICALITY AND KINETICS STUDIES

The principal purpose of this section is to indicate where basic information for physics calculations may be found. Except where some unusual or important feature needs calling attention to, only reference citations are given here.

A. Nuclides and Materials

1. Fuel, coolant

   GA-7086, p. 14.

2. Reflector, moderator

   GA-8270, p. 61.

3. Control rods

   GA-8420.

4. Incoloy shock absorbers

   GA-9261, p. 18.

5. Reserve shutdown, poison rods

   GA-9261.

6. Fission products

   FSAR 3.5, p. 47.

   GA-A-10850, Table 2.2.

7. Impurities

   GA-10010, p. 20 and Table 2.3.

   GA-7634, Table 2.9.

   GA-10202, p. 10.

B. Dimensions and Geometry

1. Core

   GA-9130, p. 30.

   GA-8038, p. 77.

   FSAR, Tables 3.1-1, 3.3-2, Figs. 3.1-2, 3.1-3, and p. 3.3-1.

2. Fuel particles

   FSAR, p. 3.4-2 and Fig. A.1-8.

   GA-10444.

   GA-8725, Table 2.6.

   GA-8600, Table 2.1 — Reference Design No. 6.

   GA-7453 mentions the "filler particle."

3. Fuel elements

   FSAR, p. 3.4-1, Figs. 3.4-1 and 3.4-4.

   GA-7634, p. 36.

   GA-7453, Table 2.1 (Design No. 4).

4. Control element

   FSAR, Figs. 3.4-2 and 3.4-3.

   GA-10444, Fig. 2.1 (Control element in buffer zone).

5. Reflector blocks

   FSAR, p. 3.4, Figs. 3.4-5, 3.4-6 and 3.4-7.

6. Control rods

FSAR, Figs. 3.5-4 and 3.8-4; Tables 3.5-7 and 3.5-8; p. 3.5-7.
GA-8270.
GA-8038, Table 2.9.

7. Poison rods

FSAR, Fig. 3.5-3 and p. 3.5-13.
GA-A12559, p. 2.
GA-A12447, Fig. 2.2 and Table 2.4.
GA-A10850, Table 2.1.
GA-A10754, Table 2.8.
GA-9875, Fig. 2.3.

8. Reserve shutdown

FSAR, p. 3.5-13
GA-7086, p. 10.

C. Material Content and Nuclide Concentrations

1. Fuel

GA-A12477, p. 2.
GA-A12200
Fig. 2.1. Regions refueled and reloads 1 through 6.
Table 2.1. As-built initial compositions.
Table 2.2. Initial fuel load.
Table 2.3. Reload loadings for an 8-yr period.
Table 2.4. Axial zoning initially and for reload number 1.
FSAR
Table 3.5-1. Total fuel load initially and equilibrium.
Table 3.5-2. Fuel loading for reference design, initial core.
Table 3.5-3. Refueling sequence for a 6-yr cycle.
Fig. 3.4-8. Side view of the core and location identification system.

GA-A10754

Table 2.9. Reference design 12, 6-yr cycle reload data.

Fig. 2.6. Core region refueled in reloads 1 through 6.

GA-10560

Table 2.1. Material constituents obtained from chemical analysis of a composition number 2 fuel rod.

Table 2.3. Adjusted fuel loading, initial core (reference design number 12).

Fig. 2.2. Initial core fuel loading distribution.

GA-9261, p. 20. Isotopic content of uranium.

FSAR

Fig. 3.5-1. Initial core fuel loading distribution.

Page 3.3-1. 93.15% fuel enrichment used.

Page 3.5-4. Identification system top axial zones are odd numbered and bottom zones are even numbered.

GA-10444. Buffer zone adjacent to reflector. Five rows of fuel holes of composition number 13.

Page 13. Modification of design, making all six control-rod columns next to core-reflector interface contain only fuel mixture number 13.

Some information on changes in C and Si content, uranium isotopic content, and matrix carbon density.

GA-10202. Graphite density of fuel block.

GA-10313, p. 15. Effective packing fraction for fissile and fertile particles assumed to be 0.63; bonded matrix material has a density of 0.7 g/cm^3.

2. Control rod

GA-7939. Absorber bodies contain 30 wt % boron as B_4C in a graphite matrix.

3. Reserve shutdown system boron loading

GA-9720, p. 26.

4. Fresh fuel element criticality storage

GA-10313, p. 33.

5. Core

FSAR, Table 3.3-1. Weights of core components.

D. Nuclear Reaction Data

GA-10313, p. 13.

FSAR, Tables 3.5-14, 15, 16, 18, and p. 3.5-30.

E. Dimensional Changes of Materials Under Reactor Operating Conditions

GA-8555. Data concerning neutron exposure effect on pyrolytic carbon dimensions.

GA-8888. Thermal and irradiation induced dimensional changes.

GA-10011. Temperature history and irradiation effects on dimensions of graphite.

GA-6888. Fast neutrons and graphite dimensional changes.

GAMD-8738. Empirical correlations of graphite properties.

GA-9919. Dimensional changes in graphite due to irradiation.

FSAR, p. 3.4-8. Graphite dimensional changes.

Table 3.4-1. Core graphite properties.

Fig. 3.4-10. Percent contraction versus time, axially and radially, for graphite.

GAMD-8619. Irradiation dimension change in boronated graphite.

GA-A12035 (CONF-720420-8). Irradiation dimension change in boronated graphite.

F. Heat Transfer Data

GAMD-7911. Film coefficient of heat transfer. Measurements during rise to power for Peach Bottom reactor.

GA-8025. Startup tests for Peach Bottom reactor.

V. CROSS-SECTION PREPARATION PROCEDURES FOR FEW-GROUP CALCULATIONS

A. Fuel Particle

GA-7077. Calculation of resonance absorption in doubly heterogeneous media incorporated into GAROL.

GA-8879. Grain effects mentioned as not having been considered before.

GA-9130. Thorium resonance cross-section sensitivity study discussed.

GAMD-10134. Grain shielding factors evaluated.

GA-10444. Discussion of thermal flux shielding for fuel particle.

B. Fuel Rods and Reflector

GA-A10754, p. 34. Effect of end caps on reactivity shown in Table 2.10.

GA-7453. Methods for calculating Dancoff factor compared.

GA-10313. Dancoff factor of 0.42 obtained by direct numerical integration of the moderator transmission probability taking into account irregularity of fuel-rod array including control hole and hexagonal block edges. The GAM-GAROL section of GGC-5 calculates fast cross sections and GATHER portion calculates thermal cross sections. Few-group cross sections computed for various values of fuel temperature and thorium density in the fast range and for various values of carbon-to-uranium ratio and moderator temperature in the thermal range. Interpolation can then be used to obtain intermediate points. The effect of ^{233}U and ^{238}U resonance structure upon the thorium resource integral explicitly taken into account.

GA-10202. Indicates correction of Dancoff coefficient.

GA-9130. Asserts that while the effective cross sections of most nuclides depends only on the energy spectrum, ^{232}Th is a major exception requiring accounting for self-shielding for both fuel rod lumping and grain structure. Scattering kernels for graphite were obtained with GASKET and HEXSCAT. Reflector cross sections obtained with 26-group GAZE calculations.

VI. CALCULATIONS RELATED TO NEUTRONIC SAFETY ANALYSIS

A. Temperature Coefficients of Reactivity

FSAR, Table 3.5-9, Figs. 3.5-13 and 3.5-14.

GA-9875, p. 11 ff. Coefficients given for 4-yr and 6-yr equilibrium cycles at the time of reference design 9.

GA-9440, p. 28. The same at the time of reference design 7.

GA-7086. Figures 2.5 and 2.6 show temperature coefficients versus temperature from below 500°K to nearly 3000°K for an unstated reference design at the beginning of cycle and for the equilibrium cycle.

GA-A10754. Describes results of one-dimensional radial TEMCO calculation to predict total reactivity change at various times during the initial core cycle. Adequacy of one-dimensional model checked by comparison with other codes BUGTRI, GAMBLE, -RZ, and SCANAL, to obtain multidimensional results. No control-rod insertion. Reference design 12.

GA-A12030. Results for temperature coefficients from 80°F to 1700°F for a variety of rod configurations from fully inserted to withdrawn using TEMCO (one-dimensional code). For effect of xenon, the one-dimensional FEVER code was used.

FSAR, p. 3.5-98 and Table 3.5-30. Compare experiments with calculations.

GA-8025. Physics startup tests for Peach Bottom reactor; see also GAMD-7357 in which calculations are compared with experiments; GAMD-7358 for Peach Bottom measurement descriptions; GAMD-7359 where zero power noise analysis and a scram test are used for measuring temperature coefficient at Peach Bottom.

GA-8270. The uncertainty in the temperature coefficient estimated to be about 7×10^{-6}/°C over the entire temperature range.

B. Reactivity Worths of Various Materials (Pa, Sm, Xe, He, H_2O)

GA-A10754. Describes methods and results for design number 12 and summarized in Table 2.2. Figure 2.2 shows buildup of ^{233}U, ^{233}Pa, ^{149}Sm, ^{151}Xe concentrations. Figure 2.3 shows reactivity represented by these versus time as well as excess reactivity and control poison as rods and burnable poison. Figure 2.4 shows ^{149}Sm and ^{135}Xe reactivity versus effective core temperature. GAUGE 4-group and TEMCO 4 – and 7-group results compared.

GA-A12200. Xenon reactivity worths calculated for use during rise to power program. The one-dimensional FEVER code was used with three fast and four thermal groups. Partially rodded configurations are mocked up by adjusting control-rod boron density to obtain agreement with a two-dimensional calculation. Beginning-of-cycle calculations used for all results. Reactivity worth of xenon plotted as a function of power and temperature (Fig. 2.5), as a function of rod insertion (Fig. 2.6). Build-up and decay of xenon worth also shown.

FSAR, p. 3.5-48 and Table 3.5-27. Gives comparisons of calculations and experiments in Peach Bottom.

GA-9875. States that complete helium inventory is worth 1.45×10^{-4}.

GAMD-7910. Gives a comparison of calculation and experiment for xenon worth in the Peach Bottom reactor.

GAMD-7356. Gives results of measurements of helium reactivity effects in Peach Bottom caused by flow.

GA-8500, Fig. 2.7. Shows reactivity addition due to steam addition to core. The worth is 2.2×10^{-5} $\Delta\rho$/lb of water. Calculations done for core composition corresponding to one year of operation. The worth of water asserted to be less in subsequent cycles because of reduced thorium load.

C. Control Rod Worth

GA-7086, p. 70. Refers to 2-D multigroup transport calculations for mutual shadowing of rod pairs. Asserts that flux shielding factors for a single rod and for a rod pair are essentially the same. Figure 2.4 shows fractional worth of one rod pair, total control system and of scram system.

GA-7634, p. 39. The assertion is made that the effects of a partially inserted rod can be treated in a two-dimensional calculation by preserving the worth of the partially inserted rod.

GA-8038. Table 2-10 gives results of a multigroup transport calculation of rod worths using nine groups (four thermal).

GA-8270. Tables 2.4, 2.5 and 2.6 give rod worths for various rod configurations for an unstated reference design. Table 2.3 gives rod withdrawal programs. (This appears to have undergone changes.)

GA-8420, p. 83. Discusses heat generation in the rod.

GA-8879. Discusses lumped burnable poison design. Points out shortcomings of GAUGE code which smears control rod over a fuel element. Asserts that GAMTRI code allows a considerably finer resolution of core geometry. Two-dimensional transport-theory treatments are discussed; diffusion calculations with GAMTRI in which boundary conditions and shielding factors obtained from transport calculations are described.

GA-9876. Asserts that method of smearing of poison over whole fuel volume is adequate for static and depletion calculations concerning lumped poisons. Reviews previous work concerning control-rod calculations investigating adequacy of assumptions which neglect top and bottom reflectors and axial fuel variations.

GA-A10010. Compares treatments of GAUGE, SCANAL, BUGTRI-GAMBLE.

The following reports are concerned with errors and the sensitivity of control-rod worth calculations to various factors.

GA-8038. Effect of temperature, inter-rod pair distance, fuel composition, gaps, and steel canning on rod worths.

GA-8270. Dependence of rod worth on temperature and upon k_∞ of various regions.

GA-8879. Table 2.5 compares results of treating the control rod as (I) a smeared region versus an explicit treatment. (The precise meaning of the "explicit treatment" or of an associated boundary conditions is not clear from the text.)

GA-8600, p. 64. Discusses the sensitivity of control-rod worth to graphite density.

GA-A10202. GATT-2, BUGTRI/FEVER, GAMBLE/RZ, SCANAL treatments compared. Additional comparisons found in GA-A10754.

FSAR, p. 3.5-40 ff and Tables 3.5-22 and 3.5-23. Compares experimental and calculated control-rod worths.

GAMD-7354 and 7368. Control-rod worths for the Peach Bottom reactor presented.

D. Burnable Poison Rod Worths

GA-8038 and GA-8600. Calculation of shielding factors on the basis of a simple formula discussed but do not say how the key constants in the formula are obtained.

GA-8879, p. 13. Discusses transport cell and diffusion theory analysis.

GA-9130, p. 42. Poison rod heating.

GA-10010, p. 24. Boron-10 cross sections averaged over core spectrum. Comparison of SCANAL and BUGTRI poison rod treatment.

GA-9875. Compares two methods of poison rod worth calculation. Material discontinuities are said to have negligible effect on worth.

GAMD-9187. Uses two-dimensional transport code 2DF to calculate flux shield factor for poison rods. Effect of smearing poison rod on control-rod effectiveness.

FSAR, Table 3.5-25. Rod worths.

E. Peaking Factors

FSAR, Fig. 3.5-11. Local peaking factors due to unfueled graphite. Figure 3.5-12 shows effect of unfueled graphite on axial power distribution. Section 14.2.2.5 gives peaking factors for rod withdrawal accidents (see Fig. 14.2-1).

GA-10560. Peaking factor calculation included no temperature feedback.

GA-10313, Fig. 2.4. Maximum power density increase during rod insertion.

GA-9720. Peaking factors for rod withdrawal obtained from 2-D static diffusion calculation neglecting temperature feedback. Average centerline temperature of seven hottest elements.

GA-9261. Power peaking due to unfueled gaps.

F. Kinetic Parameters

FSAR, Table 3.5-10. Lists prompt neutron lifetime, effective delayed neutron fractions, delayed neutron decay constants and delayed neutron fractions.

GA-7086, Table 2.8. Kinetics parameters given for various cores.

GA-8600, p. 48. Describes effect of SiC in TRISO particles on certain kinetics parameters.

GA-A12559. Describes transient parameter calculation using BUGTRI-GAMBLE.

G. Scram Reactivity Analysis

FSAR, 14.2.2.3. Describes scram reactivity assumptions and cites typical examples. Scram reactivity is said to be sufficient to leave core at least 0.01 Δk subcritical with due regard for possible rod pairs out of service and for subsequent reductions in temperature.

FSAR, 3.5.6.2. States at least 0.2-sec scram delay time assumed in transient calculations.

FSAR, 14.2.2.1. Describes protective action taken by control system to minimize consequences of a rod withdrawal accident.

GA-9875 and FSAR, 14.2.2.2. Control rod reactivity addition rate discussed. See Also GA-7314. For effects of different scram times, see GA-7086 and FSAR 3.5.3.

H. Transient Accident Analysis

Much of the information falling under this heading has already been discussed in Section IIA above, and will not be repeated here except to cite the reference number.

1. General

FSAR, Sect. 3.5.6.2. Assumptions underlying kinetics analysis listed. See also GA-8676.

2. Rod accidents

GA-8600. The sensitivity of calculational results to Λ discussed.

GA-A-12977. Rod worths tabulated in Table 2.3.

GA-12325. Control rod withdrawal sequence and rod worth listed in Table 2.8.

GA-B10872. Topical report on maximum rod worth and rod withdrawal accidents from 2% to 100% power.

GA-10980. Table 2.4 displays initial conditions and transient results; peak temperatures appear to be found by extrapolation.

GA-10560, p. 40. Table 2.10 displays results of analysis for rod withdrawal accident from source power.

GA-9720. Presents results of calculations for rod withdrawal accidents for Δk = 0.01, 0.016, and 0.025.

FSAR, p. 14.2-10. Discussion of maximum worth control rod pair withdrawal at full power; page 14.2-12: discussion of maximum worth control-rod pair withdrawal at source power with partial coolant flow; page 14.2-13: discussion of simultaneous withdrawal of all 37-rod pairs.

See also the following: GA-7086, p. 22; GA-7314, p. 17; GA-8038, p. 86; GA-8270; GA-8420, Table 2.3; GA-8600; GA-9720, p. 22; GA-9875; GA-10560; GA-10850; GA-A12477.

I. Xenon Oscillation Studies

GA-8420, p. 67. Discusses axial xenon stability.

GAMD-7213. Presentation of linear analysis of xenon instability.

VII. MISCELLANEOUS INFORMATION

A. Radiation Damage

GA-10468. Reports performance of BISO and TRISO coated particles in 275 tests at burnups up to 75% FIMA and fluences up to 8.7×10^{21} neutrons/cm^2 (E > 0.18 MeV) and temperatures up to 1325°C. Fuel rods and reference needle coke also tested.

GAMD-2361 (Parts 25, 26, 27, 28, 29, 30). Describes coated particle development for FSV and irradiation tests. Analytical studies of coated particle mechanical integrity reported. States that test conditions for irradiation were equivalent or beyond peak FSV conditions. (These reports also include fuel studies relevant to fast reactors.)

GAMD-7377. Series of tests of exposure of candidate core plate and thermal insulation materials to impure helium at 1650°F to 1850°F for 3000 hr. Results showed 430 stainless steel best choice as metallic thermal insulation material in an HTGR. Hastelloy X least affected by HTGR environment but considered prohibitively expensive in foil thickness materials.

GA-10099. Describes Charpy impact test results of irradiation-damage surveillance program on reactor vessel steel carried out in Peach Bottom reactor.

B. Protective Actions by Control System in Rod Withdrawal Accidents

FSAR, pp. 14.2-5 ff. Lists protective actions taken to limit consequences of rod withdrawal accidents. Conditions for withdrawal prohibits described, as well as automatic and manual scrams and manual insertion of reserve shutdown system.

GA-7453, p. 58. Gives lines of defense in a rod withdrawal accident. Discussion is similar to that in FSAR.

VIII. GLOSSARY OF COMPUTER PROGRAMS USED BY GENERAL ATOMIC COMPANY FOR NUCLEAR DESIGN

The most recent document on this subject, which we specially commend to the reader is Gulf-GA-A12652, "Nuclear Design Methods and Experimental Data in Use at Gulf General Atomic" by Meldon H. Merrill, July 1973.

A. Neutron Distribution, Kinetics, and Depletion Computer Programs

Code name	Report No.	Description
BLOOST-6	GAMD-8119, GA-8416	Point kinetics with thermal model of coated particle. Time-dependent, 2-D, heat transfer code with spherical code for particle.
BUG-2/BUGTRI	GA-8272	2-D burnup code, rectangular and hexagonal geometry.
DCALC	GA-8286	Subroutine for 1-D diffusion shared by FEVER-7, GASP-7, and TEMCO-7.
DTF-IV	LA-3373	Multigroup transport code with anisotropic scattering.
FEVER	GA-2749	1-D diffusion code.
FEVER/M1	GA-9780	1-D diffusion-depletion code.
GAKIN	GA-7453	1-D multigroup, diffusion, kinetics calculation.
GAMBLE-5	GA-8188	Multigroup, diffusion, 2-D code with arbitrary group scattering.
GAMTRI	GA-9201	Triangular geometry, multigroup diffusion code. Arbitrary group scattering. Unit mesh = half side of a hexagon.
GAPOTKIN	GA-8204	Point kinetics code with general reactivity function.
GARGOYLE-II	GA-9477	Infinite-medium fuel-cycle analysis code with fuel and poison searches.
GAROL	GA-6637	Calculates effective resonance cross section including overlap effects.
GASKET	GA-7417	Thermal neutron scattering kernels calculated.
GATT/GATT-2	GA-8547	3-D, few-group, diffusion code with hexagonal mesh.

Code name	Report No.	Description
GAUGE	GA-8307 (See also GA-8879, GA-9261, GA-10010, GA-A10754)	2-D, few group, diffusion, depletion code with uniform triangular mesh.
GAZE-2	GA-3152	1-D, multigroup, diffusion code. Includes criticality searches.
GGC-5	GA-8871	Multigroup cross-section code combining GAM and GATHER.
HEXSCAT	GA-6026	Calculates coherent elastic scattering of neutrons in hexagonal lattices.
SCANAL	GA-9423 (See also GA-9261, GA-9130 and GA-10010)	Single channel synthesis depletion code.
TAP	GAMD-7248	Transient analysis of HTGR power plant performance.
TEMCO	GA-6734	Calculates temperature coefficient.

B. Miscellaneous Other Codes

Code name	Report No.	Description
FREVAP-8	GAMD-8476	Calculates release of metallic fission products.
FREVAP-9	GAMD-8813	Calculates release of metallic fission products from HTGR cores.
GALAHAD	GA-9166	Code for optimization of control of xenon transients.
NEFIRS	GA-8069	1-D survey calculation of flux in shields.
TAMER	GAMD-7397	Calculates flow and temperature distribution for coolant and fuel element during transient including natural and forced convection.

C. Program Comparisons

GA-10202. Describes test of GATT-2 using cross sections from 7-group BUGTRI calculation. Compared GATT-2 transverse distribution with BUGTRI-GAMBLE-RZ-SCANAL. The axial distribution is compared with SCANAL-BUGTRI-FEVER.

GA-8879. Synthesis checked against 2-D R-Z calculations.

GA-9261. GAUGE, SCANAL compared with GAMTRI. Use of a finer mesh investigated and multithermal group effects studied.

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6830	July 1965-September 1965	9720	July 1969-September 1969
6950	October 1965-December 1965	9875	October 1969-December 1969
7086	January 1966-March 1966	10010	January 1970-March 1970
7314	April 1966-June 1966	10202	April 1970-June 1970
7453	July 1966-September 1966	10313	July 1970-September 1970
7634	October 1966-December 1966	10444	October 1970-December 1970
7939	January 1967-March 1967	10560	January 1971-March 1971
8038	April 1967-June 1967	A10754	April 1971-June 1971
8270	July 1967-September 1967	A10850	July 1971-September 1971
8420	October 1967-December 1967	A10980	October 1971-December 1971
8600	January 1968-March 1968	A12030	January 1972-March 1972
8725	April 1968-June 1968	A12200	April 1972-June 1972
8879	July 1968-September 1968	A12325	July 1972-September 1972
9130	October 1968-December 1968	A12477	October 1972-December 1972
9261	January 1969-March 1969	A12559	January 1973-March 1973
9440	April 1969-June 1969		

Only the Part I, Task II section was reviewed. This is the Nuclear Analysis Section.

<u>Topical Reports</u>

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APPENDIX

THE STRUCTURE OF ACCIDENT ANALYSIS STUDIES FOR HIGH-TEMPERATURE GAS-COOLED REACTORS

As part of the foregoing review, an outline of the overall nature and structure of accident analysis studies was constructed as a guide. A total picture is presented of the relationships between the accident scenario, the methods of accident calculation, the use of physical data, and the ultimate goal of public protection. It is not claimed that the analysis described here involves any original method. Rather, it is presented to aid the reader in obtaining an overall picture.

I. GENERAL REMARKS

As shown in Fig. 1, an accident analysis may be conceived of as a scenario introduced into a calculational procedure. The latter starts with raw data and embodies a processing method for converting the data into a form usable in the accident calculation which may consist of anything from a pencil-and-paper operation to the use of a collection of computer codes organized in modular fashion. We shall discuss below the principal features of each of these parts for the diagram with special reference to the needs of HTGR safety studies.

II. ACCIDENT SCENARIOS

In a nuclear power station the fuel is the primary source of radioactivity. (In the case of the Fort St. Vrain reactor, fuel is regularly located in the core itself, the storage wells, temporary storage areas, and the fuel handling machine.) One is, therefore, led to construct sequences of events — scenarios — which may lead to release of radiation, in order to seek out ways of preventing their occurrence. These scenarios are easily the most controversial aspect of safety analysis for they deal with probabilities rather than certainties. If something "can" happen — that is, if a sequence of events does not involve a violation of natural

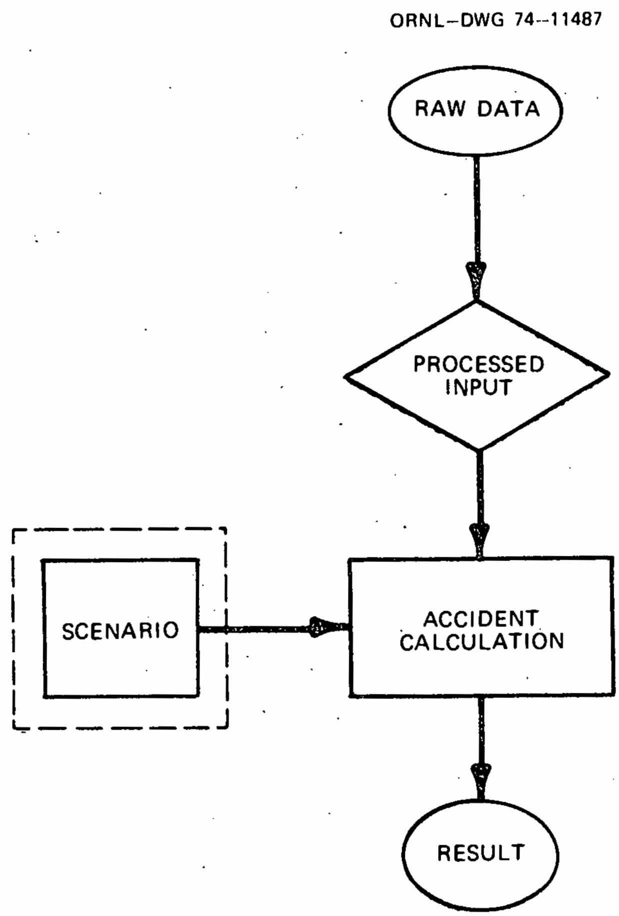

Fig. 1. Safety analysis structure.

laws — then it "will" happen — that is, there is a finite probability that the sequence will occur. It is necessary to place these probabilities in order with respect to one another. A way of doing this is suggested by reference to Fig. 2. Every node in this figure has a calculational procedure and a sequence of events associated with it which could be used to produce a graph of probabilitity versus extent of failure. Proceeding from level to level, the final result from the chart would enable the probability of any amount of radioactivity release to be ascertained. By assigning a cost to a release magnitude weighted by its probability, the hypothesized scenarios can be ordered numerically and the system reliability assessed.

Figures 3 and 4 enlarge the accident scenario sequence box of Fig. 1 to show the parts to be considered. The initial phase is where the perturbation to the normally operating system is made. The reactor will always be in a geometrically integral condition at the start of this phase. In relation to Fig. 2, this phase is the lowest level of the chart. The protective phase occurs when the protective action is taken. The other phases are self-explanatory. It has to be realized that through its lifetime the reactor system undergoes many stages of operation, each with its own special features. These various stages are:

1) initial load,
2) initial startup (including tests),
3) normal on-line operations,
4) refueling, service lifetime
5) maintenance, tests and inspections
6) mothballing process,
7) decay period,
8) final disposal.

Each of these have to be considered in the accident analysis and in their relation to the overall scheme.

In this report we concern ourselves with nuclear driven accidents in the Fort St. Vrain reactor during its service lifetime. The classes of material which are of neutronic significance are: coolant, fuel, reflector, moderator, water-steam, control rods, poison rods, structural material, fission products, reserve control spheres, and impurities.

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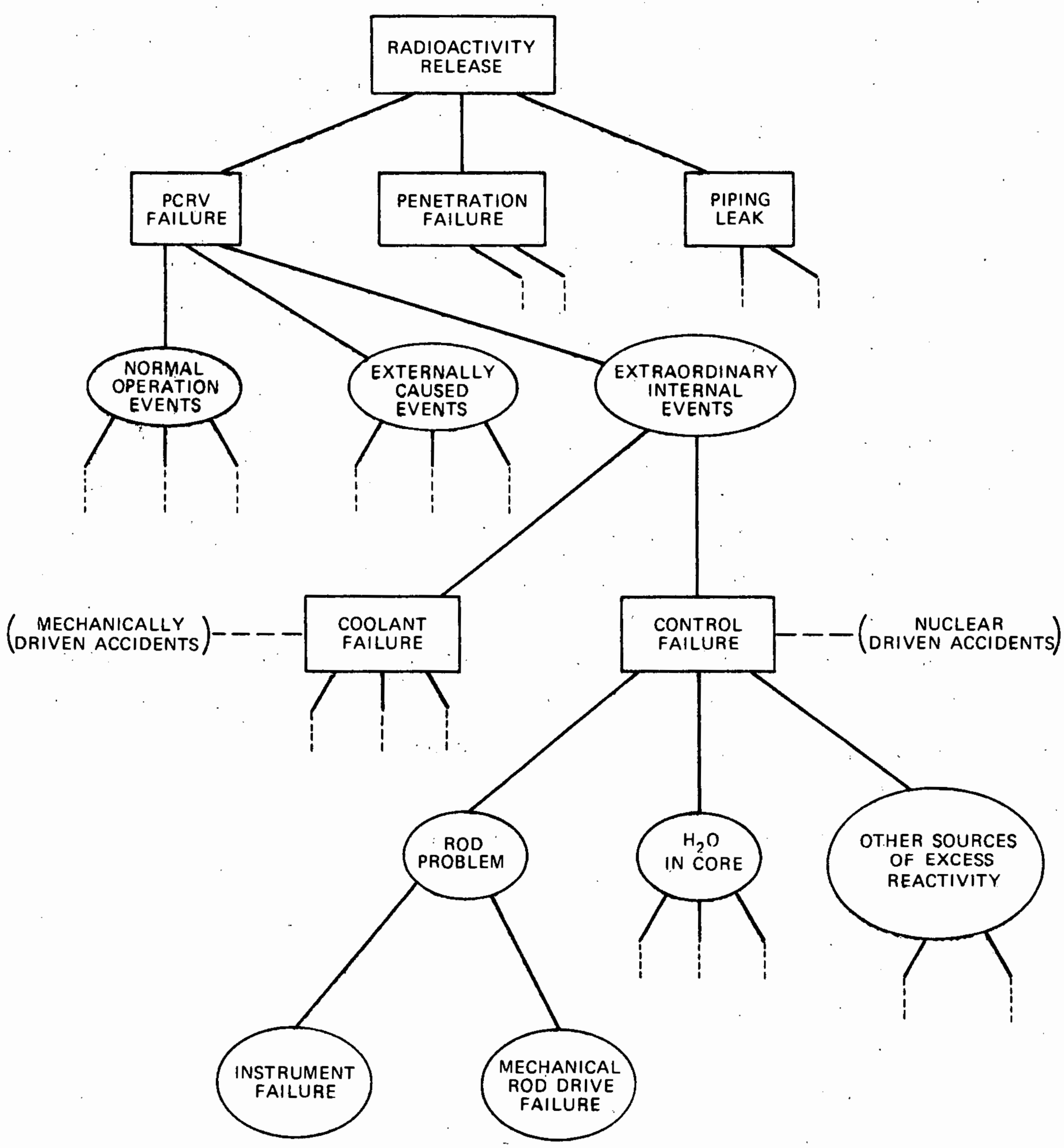

Fig. 2. Hypothetical reactor failure modes.

ORNL-DWG 74-11489

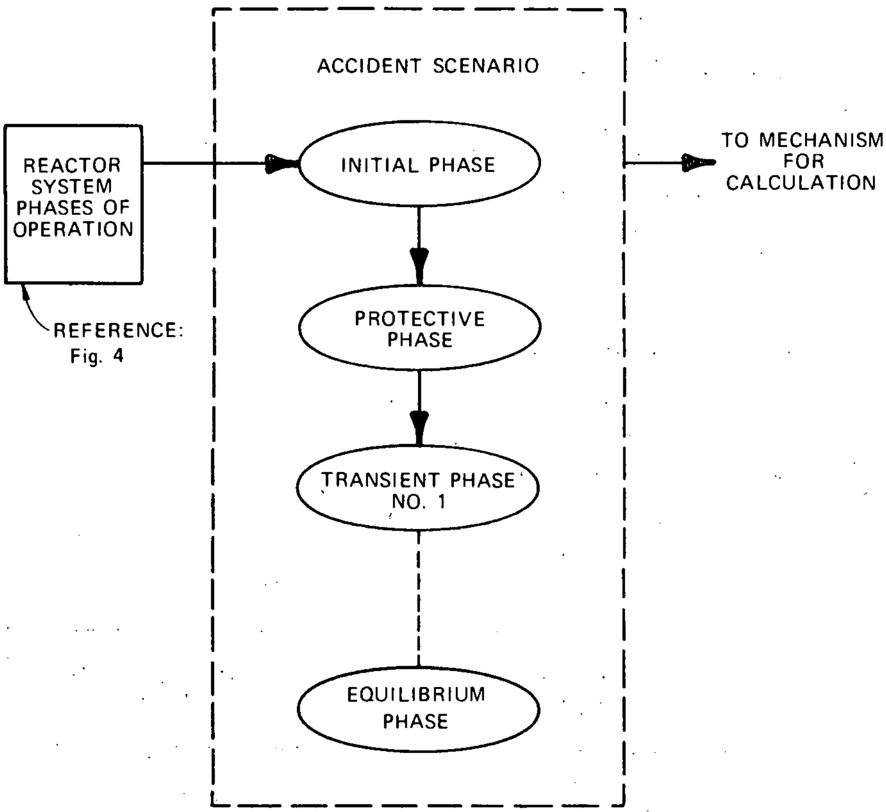

Fig. 3. Accident scenario — detailed.

ORNL-DWG 74-11490

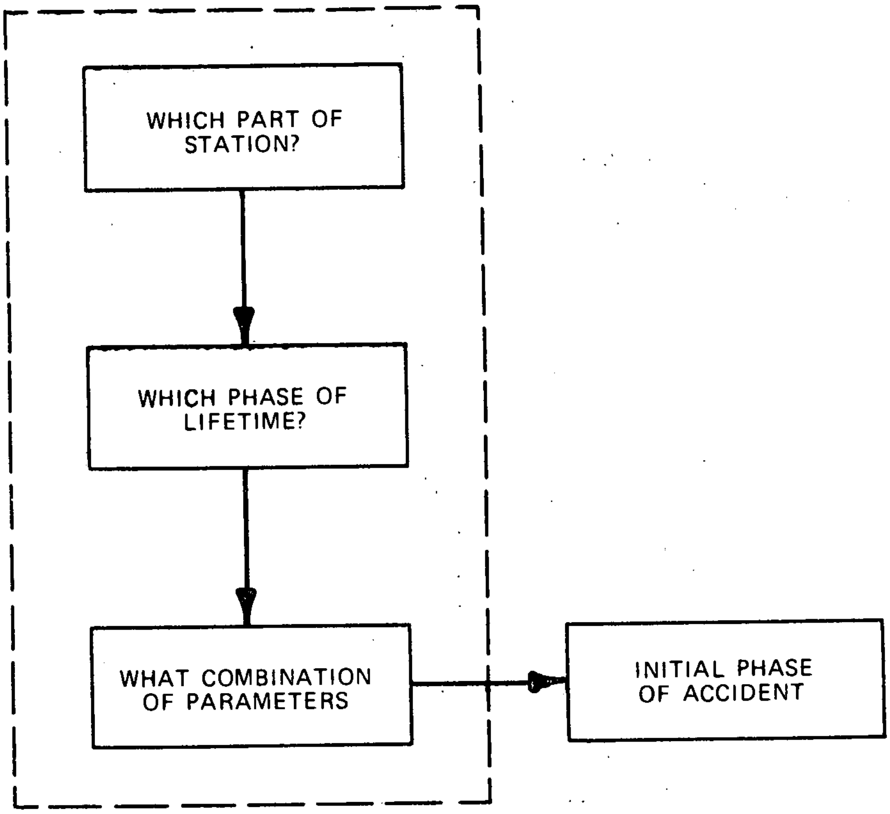

Fig. 4. Initial phase — detailed.

The fuel handling machine, temporary storage areas, and the storage well use no control rods; sand is used in connection with the storage wells.

The state of the system can be characterized by the following parameters:

1) position and dimension of each of the above materials,

2) quantity of the material,

3) the operating conditions: temperature, pressure, and power level.

The various possible combinations of these parameters constitute the initial phase of the accident scenario.

III. ANALYSIS OF NUCLEAR DRIVEN ACCIDENTS

The pattern of accident calculation is outlined below:

1. Collection of data.

For those nuclides and materials to be considered, the following information is needed.

(a) Basic nuclear data (cross sections, delayed neutron fractions, yields, decay constants, etc.).

(b) Basic non-nuclear data: densities, weights, and dimensions:

Dimensional change properties

Thermal coefficients of expansion

Irradiation dimensional effect data

Compressibility coefficients

Heat transfer characteristics

Specific heats

Conductivities

Heat transfer correlations

2. Analysis of data to estimate error bounds.

3. Calculations common to all neutronic analyses.

Calculation of kinetic parameters: prompt neutron lifetimes Λ, effective delayed neutron fractions β_i, and temperature feedback coefficients.

4. Accident scenario.

5. Static calculation.

6. Kinetic calculation.

In those cases where there is no control system available, the principal method of accident analysis will be the static calculation of reactivity. Where there is a control system available, the static calculation will merely be the preliminary calculation for accidents for which reactivity insertion is the driving force. It will provide the rates needed as input to the kinetic calculation which forms the main portion of the analysis of such accidents. Both (5) and (6) can be further subdivided into two parts:

(a) Nuclear calculation — flux is the state variable.

(b) Mechanical — a thermal-hydraulic analysis.

When temperature and pressure are the state variables for nuclear driven accidents, this may be regarded as feedback into the nuclear part.

The following matters must be considered regarding the method of solution whether one is setting up or evaluating a computational method.

Analytical — Assumptions involved in the mathematical modeling of the situation. In the case of the nuclear part:

(1) Which equations are to be solved? For, example, the use of transport theory versus diffusion theory.

(2) How is the relationship between the space, time, and energy variables treated?

(3) How are those variables treated by themselves?

(4) What are the appropriate boundary conditions to be used?

Numerical — The codes used, numerical errors, limitations and uncertainties.

7. Interpretation of results.

The behavior of the state variables (flux, temperature, and pressure), as a function of time obtained will now have to be translated into safety-related physical consequences. As far as the reactor is concerned the areas of interest are:

(a) Containment: damage to PCRV, moderator, fuel, and structural material.

(b) Control: damage to control rods, rod drives, instrumentation, and reserve shutdown system.

(c) Coolant: damage to steam generators, circulators, purification system, and metal structures.

The interpretation of the results will then require a knowledge of the effects of temperature, pressure, and flux on the components detailed above.

All these areas (1 through 7) have been synthesized and condensed into Fig. 5. But, before the diagram is discussed, one more area of importance should be mentioned. This is the question of sensitivity studies. As methods are always approximations and data always contain errors, sensitivity studies will give an idea of how important those variations are. Each rectangle in Fig. 5 can be examined by perturbing both the process and the inputs, and the corresponding output changes measure the importance of certain procedures or parameters.

Figure 5 is an elaboration of the vertical flow depicted in Fig. 1. Some details will now be discussed.

In Fig. 5 the circles indicate data, the rectangles calculational or other logic procedures, the diamonds indicate calculated input and the ellipses contain the final numerical results. The solid lines indicate a direct connection with the 'nuclear' portion of the analysis, while the dashed lines are connections to the 'mechanical' part.

The homogenization procedure referred to in Fig. 5 can be further subdivided into three parts, as shown in Fig. 6.

(a) Treatment of particles. This has to take into account fuel and coating.

(b) Treatment of fuel rod. The features of importance are the gaps above the rod, the homogenized fuel particle, the filler, and the carbon ends.

(c) Treatment of element block. This must account for homogenized fuel rod, the graphite moderator, the helium holes, the poison rods, the control rods, and the reserve shutdown balls.

Particular attention has to be paid to the way in which the poison and the control material are handled.

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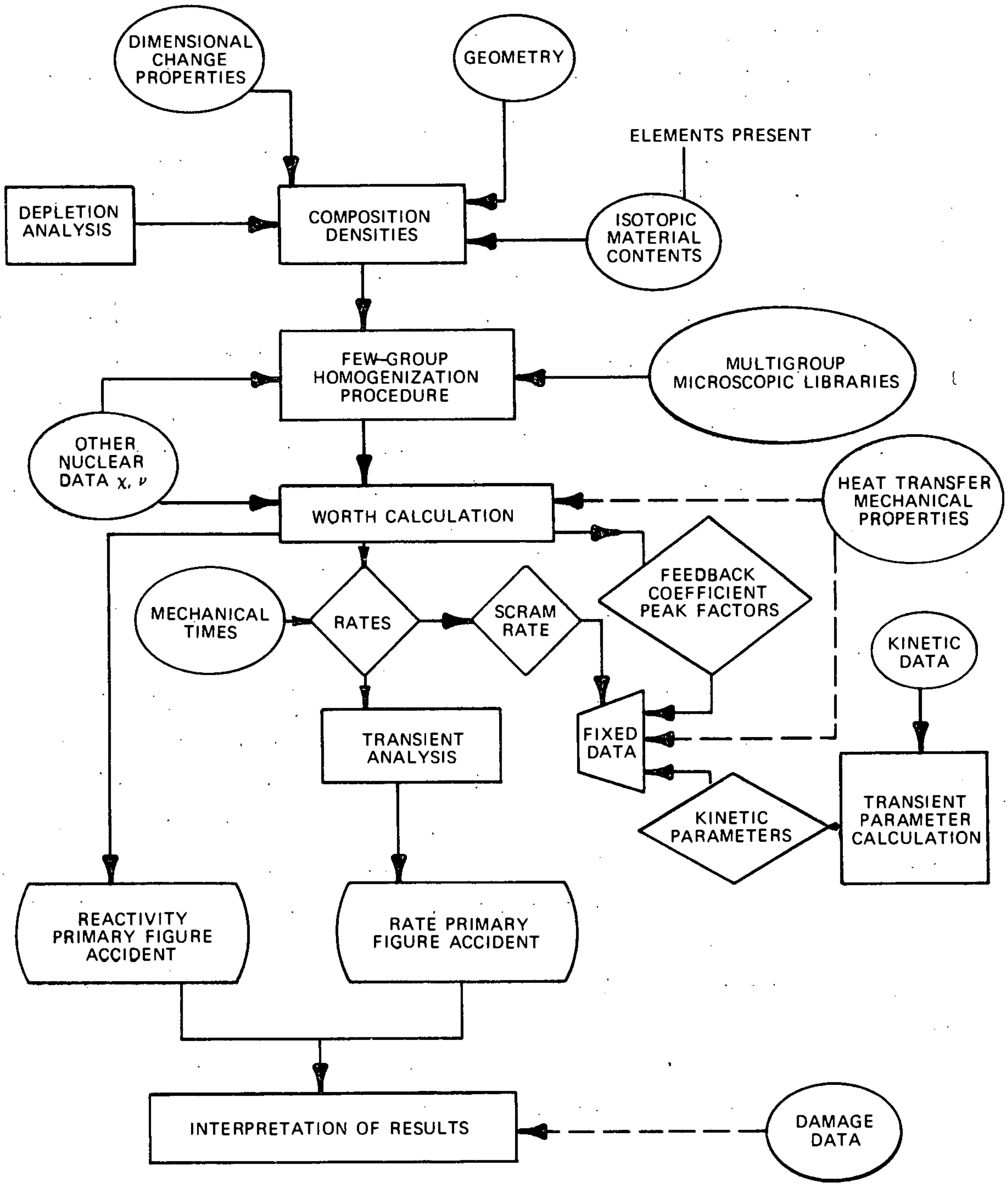

Fig. 5. Detailed diagram of calculational procedure.

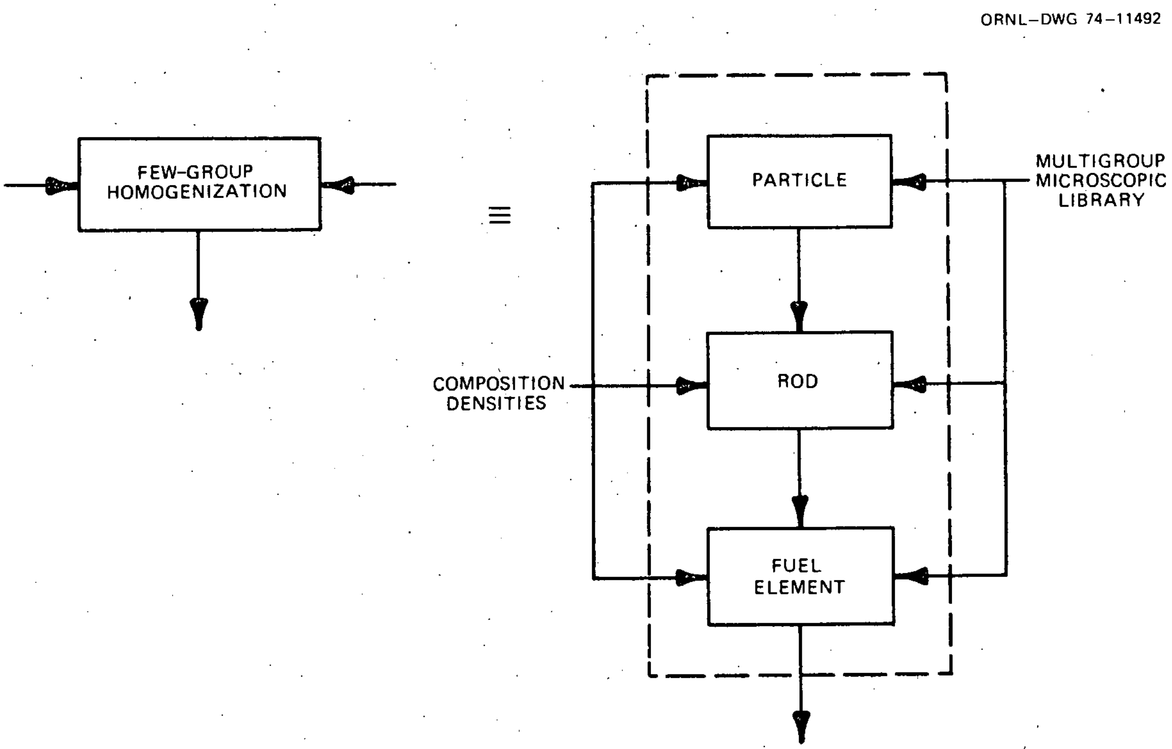

Fig. 6. Homogenization procedure – detailed.

The worth calculation and the transient accident analysis node may be similarly elaborated, as in Fig. 7.

The worths to be calculated fall under two headings. First are those worths which are fairly standard to calculate as they are required for the simplification of neutronic calculational procedures. These are worths such as xenon worth, control-rod worth, worths relating to temperature coefficients. Second are the worths which determine the accident being analyzed. The worth of water (steam) is an example. Control-rod worth can also be included in this group. Others are fission-product worths and fuel worths. What is to be included in this section has to be decided by what the initial phases of the accident scenario are possible.

In a similar way, for transient accident analysis the physical condition which feeds into these calculations is defined by the configuration and the composition density of each material, the temperature, pressure and power level. These come from the accident scenario. Figure 7 illustrates this relationship.

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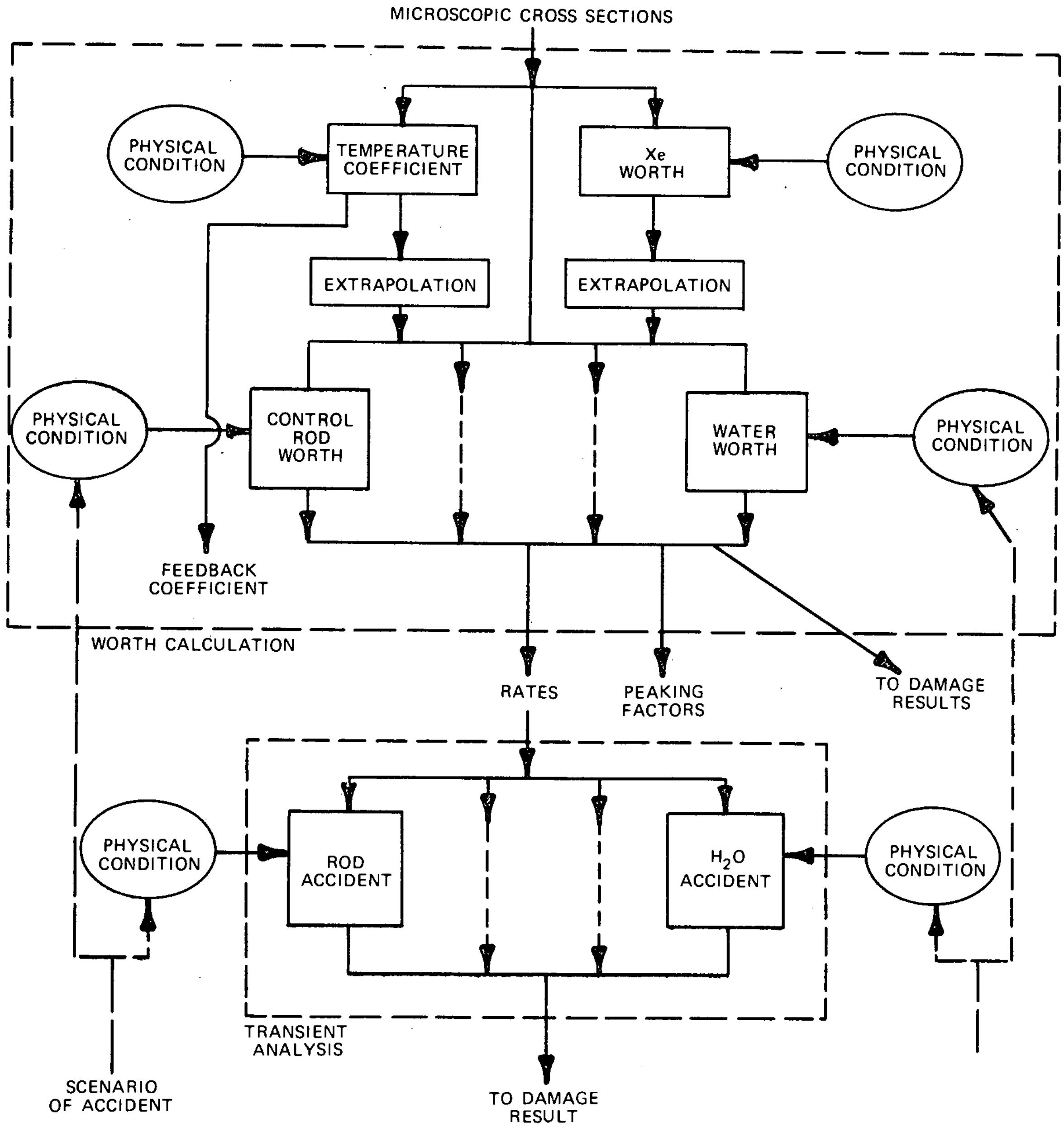

Fig. 7. Worth calculation and transient analysis nodes of Fig. 5 elaborated.

ORNL-TM-4774
(UC-77 Gas-Cooled
Reactor Technology)

Internal Distribution

1. T. E. Cole
2. J. H. Coobs
3. W. B. Cottrell
4. F. L. Culler
5. F. E. Dyer
6. J. R. Engel
7. G. G. Fee
8. M. H. Fontana
9. A. P. Fraas
10. Uri Gat
11. H. W. Hoffman
12. F. J. Homan
13. J. D. Jenkins
14-51. P. R. Kasten
52. R. J. Kedl
53. T. B. Lindemer
54. A. L. Lotts
55. R. N. Lyon
56. R. E. MacPherson
57. A. P. Malinauskas
58. A. M. Perry
59. H. Postma
60. J. P. Sanders
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